Peter Zarakas
Vice President

Consolidated Edison Company of New York, Inc.
4 Irving Place, New York, N.Y. 10003
Telephone (212) 460-5133

September 29, 1980

RE: Indian Point Station
Unit Nos. 1 and 2
Docket Nos. 50-003 & 50-247

Mr. Boyce H. Grier, Director
Office of Inspection and Enforcement
Region I
U.S. Nuclear Regulatory Commission
631 Park Avenue
King of Prussia, Pennsylvania 19406

Dear Mr. Grier

Enclosed you will find two (2) copies of Changes, Tests and Experiments for the year 1979, as required by 10CFR50.59(b). Attachment I relates to Unit No. 2.

Indian Point No. 1 was shutdown on October 31, 1974 and is presently in the defueled condition. It has been determined that this requirement is not applicable to Indian Point Unit No. 1.

Very truly yours

Peter Zarakas

Attachment:

CC: Mr. Victor Stello, Jr., Director (40 copies)
Office of Inspection and Enforcement
c/o Distribution Services Branch, DDC, ADM
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Mr. William G. McDonald, Director (2 copies)
Office of Management Information and Program Control
c/o Distribution Services Branch, DDC, ADM
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Mr. T. Rebelowski, Resident Inspector
U.S. Nuclear Regulatory Commission
P.O. Box 38
Buchanan, N.Y. 10511

A008
S
40/40

8010200310

ATTACHMENT NO. 1

CHANGES, TESTS AND EXPERIMENTS - 1979

1. Modification to Air Dryer Filter Piping

A new filter assembly and associated piping was installed for Refrigerant Dryers #21 and #22.

The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report shall not be increased because the instrument air system's function and capabilities shall remain unchanged by this modification.

Since no new probability of rupture was introduced into the system by this modification (class A and seismic I), the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report shall not be created.

The system's parameters and operational capability shall remain unaffected by this modification so that the margin of safety as defined in the bases for any technical specification shall not be reduced.

The modification to instrument air piping inlet and outlet of the filter assemblies to #21 and #22 Instrument Air Refrigerant Dryers is deemed not to involve an unreviewed safety question.

2. Test Connection at SIS Pump No. 22

A test connection was installed on the discharge of #22 SIS pump and before the MOV's 851A & 851B. The test connection piping has a shut off valve and a tubing cap. This will enable plant test personnel to attach pressure sensing equipment to the tubing after the removal of the tube cap.

The SI system has been evaluated in section 6.2 of the Unit 2 FSAR. This modification will provide a test point for pump condition evaluation. Within section 6.2 - page 2&3 the test connections are discussed, showing that there are many similar type test point throughout the SI system. Thus, this modification will not increase the probability of occurrence in the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report.

The presence of test connections for the purpose of determining system operability, has already been addressed in section 6.2-3 & 4 of the safety analysis report. Thus, the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created.

The bases for the Unit 2 technical specifications, page 4.5-6, requires that the SIS pumps be run monthly to show that the pumps are in satisfactory running condition. This modification will allow the test requirements to be met in a timely manner. Therefore, the margin of safety as defined in the bases for any technical specifications will not be reduced. Thus this project is deemed not to involve an unreviewed safety question.

3. New Fittings for Testing FCU Charcoal Filter Dousing System Nozzles

A fitting consisting of a Swagelok type male elbow with a pipe nipple and cap was placed on each of the five lowest level nozzle headers of the fire dousing system for each Fan Cooler Unit charcoal filter plenum. These fittings serve as a drain points for any residual water which may enter the system during testing or at other times and enable personnel to connect an air hose to the nozzle headers and thereby pressurize them for the flow test.

The modification involved a passive component whose main function is to provide a means for dousing system nozzle testing. Beyond such testing, the new fittings have no effect on either the dousing system or Containment Fan Cooler Units. They cannot, therefore increase the probability or consequences of any previously evaluated accident or create the possibility of a new type of accident.

In addition, since the changes will allow us to perform nozzle tests as requied by the IP #2 technical specifications, the bases for those specifications are not adversely affected.

Therefore, it is concluded that the modification does not constitute an unreviewed safety question per 10 CFR50.59.

4. Replacement of Level Transmitter (LT 931) on Spray Additive (NaOH) Tank

The existing displacement type transmitter was replaced by a type of transmitter which is better suited to this service. The new transmitter was installed in the same location as the existing transmitter with minor piping modifications to accommodate the new instrument.

This modification provides an improved method of level indication but does not alter the function of the transmitter or affect the actual tank level. The operation of the Containment Spray System will be unaffected by this modification. No fire hazard concerns are raised and the Security Plan is unaffected by this modification.

Since the modification involves an improved method of assuring that the required amount of NaOH is available to the Containment Spray System following a postulated accident, the consequences of such an accident are in no way increased.

The probability of occurrence of previously evaluated accidents is not affected by this modification.

Since, as discussed above, the NaOH tank level and the functioning of the Containment Spray System are unaffected, no new type of accident can be created.

Finally, the changes provide further assurance that the technical specification limitation on available NaOH will be met. The margin of safety as defined in the basis for that technical specification is not, therefore, reduced.

Thus, it was determined that the modification does not consititute an unreviewed safety question per 10 CFR 50.59.

5. <u>Boron Injection Tank Flanged Tee Connection</u>

A permanent flanged Tee with associated valves and blind flange was installed which allows filling the Boron Injection Tank (BIT) without disconnecting the tank level transmitter (LT-944B) and removing pipe insulation and heat tracing.

The new Tee and valves are also heat traced.

The new pressure boundary is the fill valve backed up by a blind flange. If the valve were to leak, the leakage would be stopped by the blind flange bolted to the valve outlet. To prevent boric acid from depositing on internal surfaces, the new elements will also be heat traced. The new Tee, valves, and flange cover are designed to the same specification and quality requirements as the existing system.

The modification does not involve a fire hazard. The BIT and its associated piping have a seismic I classification; this modification will not degrade this classification.

6. Re-Routing of Line #14 to Eliminate High Stress Condition

A 1" diameter section of line 14 (RCP motor bearing oil cooler return line) downstream of the RCP motor, was re-routed.

Since the subject modification was limited to an alteration of the piping configuration, no concerns involving operational changes or malfunctions are raised. In addition, the effect of this modification on the seismic design of the existing system has been examined and found to be acceptable. All materials and installation is in accordance with applicable specifications and standards.

The proposed modifications have no effect on fire protection or plant security.

Since the modification meets all existing quality and design specifications and requires neither significant additional piping nor the installation of new components, it cannot increase the probability or consequences of any previously evaluated accident.

In addition, since the functional requirements of the line involved are unchanged and no new interface with, or effect on any safety-related equipment is introduced, no new type of accident can be created.

Finally, the plant technical specifications and their bases were not affected by this change.

It was therefore concluded that the subject modification does not constitute an unreviewed safety question per 10CFR50.59.

7. Sealing of RTD 420A

In the process of removing a failed RTD (420A) from Reactor Coolant (RC) Loop #22's hot leg, a portion of the RTD probe snapped off and remained inside the RTD boss and pipe penetration. The probe, which protruded about one inch from piping inner surface (29" inside diameter), apparently had "frozen" in place; attempts to remove it had failed. The repair included spreading the probe at the top so that it cannot fall into the system and welding a stainless steel cap on the pipe's RTD boss.

There are three RTD's located in each hot leg of the four RC loops; one of these three is used for temperature sensing while the other two are spares. The temperature signal from each hot leg is used in the Reactor Protection System (RPS) for initiating overpower and overtemperature delta T trips. For RC Loop #22's hot leg, 420A was capped off and will not be used, 422A is presently being used, and 421A is a spare. With respect to the temperature signal from a hot leg, the installed spares only provide flexibility if the operating RTD fails, since only rewiring to one of the spare RTDs is required. Therefore, reducing the amount of available spares only reduces operational flexibility and does not affect reactor safety.

The probe, which remains in the capped RTD boss, has been spread at its top, this will prevent the RTD from falling into the RC system. Even if the probe was not spread, it is highly unlikely that the probe could drop into the system (weighing about two to four ounces) in light of the large force applied to the probe which failed to remove it.

This modification does not involve a fire hazard.

Since the RPS has not been degraded, the RCS pressure boundary integrity is maintained, and the broken RTD probe cannot fall into the RC piping, the following is true:

> The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis has not been increased.
>
> The possibility for an accident or malfunction of a different type from any evaluated previously in the safety analysis report has not been created.
>
> The margin of safety as defined in the basis for any technical specifications has not been reduced.

For these reasons this modification is deemed not to be an unreviewed safety question.

8. <u>Installation of Under Voltage Alarm in 480 Volt Safeguards Bus</u>

This modification alerts the control room operator to an undervoltage condition on the 480 safeguards buses that could produce a marginal condition should safeguards equipment be required.

The modification taps off the 120V supply past the 480-120V transformer. The new undervoltage relay, 47, will monitor the voltage from the 480 volt safeguards buses and alarm in the CCR.

A separate indicator light will be used to provide indication that the undervoltage alarm circuit is activated.

The new undervoltage alarm is an addition to already existing undervoltage relays. Whereas the existing relays trip the bus at 46% of normal voltage the new relays will alarm at 92.7% but have no automatic function.

The undervoltage alarm is used for operator information. There is no automatic function associated with its operations. The alarms will not change any existing control or alarms on the 480 volt safeguards buses. Thus, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report is not increased.

Similar low voltage alarms and trips are now existing on the 480 volt system. This addition will not change or produce a condition different from that which now exists. Therefore, the possibility for an accident or malfunction of a different type from any evaluated previously in the safety analysis report is not created.

This modification will not change the technical specification or bases as described in section 3.7, but will aid in determining an undervoltage condition, either during normal operation or when the diesels are in operation. Thus, the margin of safety as defined in the bases for any technical specification is not reduced.

Therefore, this project was deemed not to involve an unreviewed safety question.

9. <u>Generic Replacement of Carbon Steel Cement Lined Service Water Piping with Stainless Steel Pipe</u>

A section of 10" fan cooler unit return line (line #12b) was replaced with stainless steel pipe, when the existing carbon steel cement lined pipe developed leaks due to corrosion.

The stainless steel material involved is capable of withstanding the harsh corrosive environment associated with river water (sodium chloride content of the water can vary from 20 to 15,000 ppm).

The stainless steel alloy used, type 304 is listed in UE & C specification 9321-01-248-35 page V-2A for service water use and is used in many parts of the existing system. This alloy has proved to be resistant to the corrosive environment of the service water system and was certified for use by Station Analytical Engineering in service water applications.

The use of braze joints and couplings is considered acceptable in that the water coils in the Fan Coolers are a brazed assembly.

The replacement piping is the same size and schedule and utilizes existing pipe hangers. The stainless steel alloy is more ductile than carbon steel cement lined pipe, thus, making the replacement pipe better from a seismic standpoint than the original installation.

The replacement of leaking carbon steel cement lined pipe with stainless steel will not degrade the service water system or operation of the Fan Coolers in any way. The new material is capable or withstanding any analyzed accident that the original pipe was designed to.

For these reasons the modification is deemed not to be an unreviewed safety question.

10. Replacement of Existing Dual Current Source & Dana Amplifiers in T avg. and Δ T Circuits with Foxboro Single R/E Converter E 695 Units

The R/E converter E694 unit replaced the dual current source (eg. 2TT-421 A & B's) and Dana Amplifier (eg. 2 TM-421 & 422 A). The E694 units will convert the resistance indicated on the cold and hot leg resistance temperature detectors (RTD's) on each loop to voltage and will amplify that voltage signal.

The Foxboro R/E converter E694 units are compatible with the existing instrumentation and their output characteristics match and fit the temperature resistance curve of the existing RTD's to obtain a 2-10 volt linear output for the range of 540°F - 615°F. The accuracy & reliability of the E694 units is better than the former dual source and Dana amplifiers so that the overall accuracy of measurement of the system will be improved by this replacement. Minor time constant adjustments on the input circuits of the E/I and E/E modules will not affect either the overall accuracy or overall time constants (from sensor to the reactor trip breaker) of the system. The installation of the E694 units will not affect the K constants defined in the technical specification 2.3.1 (4) and 2.3.1(5). This modification is not involved with either the plant security or plant fire protection plans.

The replacement of the dual source and Dana amplifiers with the Foxboro single R/E converter E694 units will improve overall measurement accuracy and will not affect the overall time constant for the system so that accident analysis in the F.S.A.R. are still valid. Therefore, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report will not be increased.

Any type of failure or malfunction of a new R/E converter resulting in erroneous measurement or loss of that circuit is considered in the circuit design of the 4 channel logic system with a 2 out of 4 channel trip signals causing the reactor trip on overtemperature and overpower. As a result, the probability for an accident or malfunction of a different type from any evaluated previously in the safety analysis report will not be created.

Since the overall accuracy of the system will be improved and this modification will not affect either the calibration, test, or K constant (technical specification 2.3.1.(4) and 2.3.1.(5) requirements) the margin of safety as defined in the basis for any technical specification will not be reduced.

Therefore, the replacement of the dual current source & Dana amplifiers in the T avg. and ΔT circuits with Foxboro single R/E converter E694 units is deemed not to involve an unreviewed safety question.

11. Install Rooftop Air Conditioning System upon Control Building

To accommodate the additional heat load added to the Central Control Room (CCR) by the operation of the Central Alarm Station (CAS) electronic equipment, an air conditioning unit was installed on the Control Building Roof.

The new system serves to cool and recirculate air for the CAS and Prodac computer areas. It does not serve to ventilate the CCR, nor supply the CCR with outside air.

The Control Building is classified as seismic I. Analysis has shown that the attachment of the evaporator and roof unit to the Building's structural members does not degrade this classification. The failure of the proposed unit will not result in an unsafe condition in possibly causing the CAS or Prodac to overheat. The CAS is not safety related with respect to safe shutdown of the reactor and maintaining it in a safe shutdown condition. Likewise, the Prodac is an operational tool, and is not essential for the safe operation of the plant. The evaporator is located between roof and hung ceiling; if it fell from its supports, it would land in an empty walk way and not interfere with the operation of any safety related equipment.

The roof of the CCR is essentially a part of the enclosing structure for the CCR. During a LOCA, the outside air is presumed to be heavily contaminated; the present ventilation system shifts to a mode where by the CCR is under a slight positive pressure, all incoming outside air is filtered, and the CCR air is filtered to reduce any particulate levels. Since it is important to maintain ventilation integrity of the CCR enclosure the roof penetrations will be airtight and waterproof.

This modification does not degrade the fire detection or suppression facilities afforded the Control Room. The fire loading of Control Room (Fire Zone 15) is low and will also not be changed by this modification. For these reasons, this modification is not deemed to be a fire hazard.

The failure of the proposed air conditioning unit will not result in the failure of any safety related equipment and the ventilation integrity of the CCR is maintained. Thus the following is true:

The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis is not increased.

The possibility for an accident or malfunction of a different type from any previously evaluated in the safety analysis report is not created.

The bases of any Technical Specification is not changed, thus the margin of safety defined in the bases is not reduced.

For these reasons the modification is deemed not to be an unreviewed safety question.

12. <u>Hydraulic Restraint Modification</u>

During the 1979 Refueling Outage modifications to the Unit #2 hydraulic restraints fell into three catagories:

(a) Modifying existing supports to allow them to accept replacement hydraulic restraints which have different physical dimensions. (Eng. Mod. Proc. #FFI-79-2-13)
(b) Correcting the stroke length of existing hydraulic restraints (Eng. Mod. Proc. #FFI-79-2-16)
(c) Relocating existing pipe supports and whip restraints to clear pipe interferences. (Eng. Mod. Proc. #FFI-79-2-17)

Since the work accomplished did not degrade the seismic classification of the system nor functionally change the system, these modifications were deemed not to be an unreviewed safety question.

13. Temporary Oil Reservoir

A large seismic restraint on 24 Steam Generator has a small leak in its oil system, so small, about 75 cu. in. per week, that is has not been localized. The sight glass on the oil reservoir reads over a level span of about 4 inches, corresponding to about 300 cubic inches of oil. Thus, the leak will cause the level to move from the top of the sight glass to the bottom in about a month. Operations convenience and restraint reliability will be improved with the addition of a large oil reservoir.

Although the modification introduces about 26 gallons of snubber oil to VC, the location is remote from possible sources of ignition and the added risk of fire is negligible.

The modification has no plant security implications.

Mechanically the additional reservoir is as reliable as the existing one and it does not increase the likelihood or severity of the consequences of any analyzed accidents. Increasing the reserve of snubber oil, improves overall reliability of the plant.

Thus, the probability for an accident or malfunction of a different type than any evaluated previously in the safety analysis will not be created nor will the probability or consequences of an accident previously evaluated be increased.

The margin of safety as defined in the bases for any technical specification is not reduced by this modification. This is true inasmuch as there is still a more than adequate supply of snubber oil available to maintain the level in the snubbers.

It is concluded that the adding of a larger reservoir for snubber oil does not involve an unreviewed safety question.

14. Redesign of Service Water Pumps

The reliability of the service water pumps was improved by changing the design of the main bearings, strengthening of the outer column pipe, and strengthening of the impellers.

The new bearings are marine type bearings with better salt water wear resistance. A tube encloses the pump shaft and bearings to prevent the silt in the river water from entering the bearings (river water is passed through a Laval separator to remove the silt and then is flushed down the tube to lubricate the bearings). The outer column pipe thickness was increased to .5 inch to strengthen it and maintain the pump aligned. The impeller material was changed from bronze to stainless steel to reduce erosion.

The basic pump design is unchanged so that the system's characteristics, parameters and capabilities were not changed. The added weight of the bearing and shaft enclosing tube and the thicker column pipe will not degrade the pump's or the service water system's seismic I capabilities. The use of stainless steel on the impellers for pumping salt water is acceptable for such low temperature applications. The modification to the pumps will result in better reliability and will not affect the present A.S.M.E. Section XI testing requirements. This modification is not involved with either the plant security or the plant fire protection plans.

This modification will not affect the Service Water System's capabilities and characteristics so that the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report will not be increased.

This modification will not degrade the service water system's seismic I capability. Therefore the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report will not be created.

15. Modify vents to reduce mass of closure; therefore, reduce tendency toward vibration induced weld cracking

The vent points listed on Tables I, II, and III of Safety Analysis NS-2-77-084 have either an isolation valve with an extending pipe, blanked off with a blind flange or, in the case of some instrument vents, a pipe plug threaded into the instrument. When vibration of the vent line occurs, a stress is placed on the weld joint where the vent line originates. This vibration induced stress would be decreased if the mass at the blind flinge could be decreased. With respect to the vent pipe plug utilized by the instruments, after repeated use the connection would be susceptible to leakage. In attempting to eliminate a leak, overtorquing of the pipe plug and possible damage to instrument threads could take place.

The problems associated with the above vents can be solved by installation of Swagelok compression fittings. These are of smaller mass than a blind flange and in the process of tightening, two opposing torques are utilized, thus not stressing the member from which the line originates.

The modifications will be in accordance with applicable codes, approved welding procedures and material and other specifications. Therefore, the integrity of the piping will not be compromised or degraded. The Swagelok compression fittings are presently utilized in vent lines on lines of the Safety Injection System. (See Safety Evaluation NS-2-76-005)

During the 1979 Refueling Outage a number of the vents were modified as described.

The function of the vent points/lines will not be changed and the probability of their failure will be reduced with this modification. Thus, the margin of safety as defined in Technical Specifications will not be reduced.

The possibility of failure of the vent points will be reduced with the modification. The type of failure is the same with or without the modification. Thus, the possibility for an accident or malfunction of a different type from any other previously evaluated in the safety analysis is not created. Also, the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis is not created.

For these reasons the modification if deemed not be an unreviewed safety question.

16. <u>Additional Alarms for Emergency Diesel Generators</u>

A new alarm was installed in the CCR, for each diesel generator, to indicate the lock-out/shutdown condition. This alarm will also be activated in case of loss of d.c. power.

At present, various abnormal conditions in any of the three (3) emergency diesel generators, activate a single "Diesel Generator Trouble" alarm. Some of these conditions cause the emergency diesel generators to trip or prevent their start.

Under the modification, the new alarms, will be activated whenever the lock-out relay or the shutdown relay is energized or there is a loss of d.c. power. The alarms will further stay energized until the lockout is reset.

The main consideration in making this evaluation was to ensure that the modification would maintain the single failure criteria for the emergency diesel generator system. Each of the three diesel generators will have a separate alarm and the new relays will be added to their individual circuits. The failure of new components in one of the diesel generators will not affect the operational capability of the other diesel generators.

Failure analysis postulating an open-circuit condition in the new circuitry shows that the modification will not affect the operational capability of the diesel generators. An open-circuit condition, in the circuit containing R1, would result in the alarms in being inoperative which is no worse than what presently exists. An open in the circuit, containing relay R2, will falsely energize the alarm, but would not affect the operation of the diesel generator or the existing circuitry.

The new additional circuitry and the associated alarms do not pose a fire hazard.

Any malfunction due to the modification which may occur in one of the diesel generators, will not increase the probability of occurrence or the consequences of any accident or malfunction of equipment important to safety previously evaluated in the safety analysis. The safety and the technical specification had evaluated the inoperability of one diesel generator.

The modification does not create any possibility for an accident or malfunction that has not been previously evaluated and maintains the margins of safety as defined in the Technical Specification.

It is therefore, determined that the modification does not constitute an unreviewed safety question.

17. <u>Reactor Trip and Bypass Breaker Control Circuit Key-Interlock Modification</u>

In order to minimize mechanical cycling of the reactor trip/bypass breaker's under voltage trip mechanism and the reactor trip breaker themselves during logic channel testing, a key-interlock switch (with associated indication and alarm circuits) was installed in each of the reactor protection logic channels. This switch will allow the reactor trip breakers to remain racked in during the testing period thereby reducing wear on the breaker's contacts. This modification will improve the reliability of the reactor trip breakers.

The results of this modification will be:

1. To leave the reactor trip breaker in the logic train being tested, 52/RTA (52/RTB), physically racked in with the breaker in the open position but electrically racked out (the breaker will not physically change positions and the undervoltage trip mechanism will remain de-energized or in the trip position).

2. The associated bypass breaker 52/RYB (52/BYA) which is racked out will not have physical cycling of its undervoltage trip mechanism.

This modification will not affect the administrative controls and electrical interlock which prevent the bypassing of both reactor trip breakers at the same time. Since the key interlock switch will be installed in the reactor trip breaker cabinet which is located in a vital area, no degradation of the plant security plan will occur. This modification is class IE so that no degradation in the reactor trip breaker control circuit's seismic I capability will occur. The only failure mode that can occur in the key-interlock circuit during normal reactor operation, going to the bypass mode, will result in a reactor trip which is a safe condition. This modification is not involved with the plant fire protection plan.

Since the only failure mode of this circuit would result in a safe condition for the plant (reactor trip), the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report will not be increased.

This modification will not degrade the seismic capability of either the reactor trip breaker or its control circuit so that the possibility for an accident or malfunction of a different type from any evaluated previously in the safety analysis report will not be created.

By reducing the wear on the reactor trip breaker, this modification will increase the breaker reliability such that the margin of safety as defined in the basis for any technical specification will not be reduced.

Therefore installation of a key-interlock circuit in the reactor trip and bypass breaker control circuit is deemed not to involve an unreviewed safety question.

18. Acoustic Sensors on the Pressurizer Safety Relief Lines

Three acoustic sensors were installed downstream of the pressurizer relief valves and upstream of the pressurizer tank (PRT).

The sensors are strapped on the three 6 in. lines (lines No. 342, 343, 344) emanating from the pressurizer. These 3 lines join at a common 12 in. header (line No. 70) leading to the PRT. Indication is located in the CCR along with the instrument racks. The acoustic sensors are environmentally acceptable (i.e., can withstand containment temperature, pressure, radiation etc.) and seismically qualified consistent with the component or system to which it is attached.

All of the pressurizer power operated relief valves (PORV) and their motor-operated block valves (MOV) have positive position indication. Temperature sensing elements are provided downstream of each safety valve. An additional temperature sensor is located in the common manifold joining the discharge of the PORV's and safety valves. The readouts for these instrumentations are in the CCR.

In addition, the PRT has temperature, pressure and high liquid level indication readouts and alarms in the CCR.

Considering the above, the acoustic sensors constitute another indication among a series of indications.

This modification did not require cutting or welding.

Therefore, it is concluded that the modification does not constitute on unreviewed safety questions.

19. Containment Isolation Reset Circuitry

This modification adds electrical relays, to the actuation circuits of the containment isolation (C.I.) valves, which will monitor the status of the circuits (energized or de-energized). Contacts from each relay are wired in series, forming a "daisy chain", which is interposed in series with the C.I. reset circuits. As long as one or more of the contacts in the "daisy chain" is open, the reset circuit cannot be activated. To have all of the contacts in the correct condition (normally closed) for reset, all of the valve operating switches must be in the "valve closed" position.

Power for the relay coils is obtained from terminal blocks in the "E" and "F" cabinets in the control room. The relays, which are Westinghouse Types BF and BFD are also located in the same cabinets.

The monitoring circuitry has been designed to maintain:

Existing separation between C.I. Trains A and B, the C.I. reset switches must be used to clear the C.I. signal and the remote/local valve actuation option.

Existing administrative procedures require that all of the manual switches be placed in the "valve closed" position prior to resetting the C.I. signal. This modification will ensure that these procedures are adhered to and will not require any changes to the present methods of operation.

The containment isolation activation is discussed in Section 5.2 of the FSAR. This modification does not effect that description because it is only involved in the clearing of a containment isolation signal and not in its activation. No failure of the new system can effect any of the safety controls that position any safety related equipment during or prior to a containment isolation signal. Thus, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety discussed in the safety analyses report will not be increased.

This modification will assure safe valve position when a containment isolation signal is cleared and will thus, prevent any inadvertent automatic valve movement. Therefore, the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report will not be created.

The present technical specifications do not address the clearing of a containment isolation signal but NPG procedures E-2, A, B, C, D, E & F do, and where required, these procedures have been modified to reflect the electrical changes. All existing procedures will still be applicable and no changes to the existing modes of operation are required.

Thus, the margin of safety as defined in the bases for any technical specifications will not be reduced by addition of this modification.

Therefore this modification is deemed not to involve an unreviewed safety question.

20. Installation of Saturation Meter in CCR

A saturation meter has been installed in the central control room (CCR) to warn the CCR operators if the temp-press relationship in the RCS is approaching saturation values.

Signals are taken downstream of isolation signals so as not to interfere with the primary function of these signals.

The signals for the saturation meters are obtained from the reactor cold leg resistance temperature detectors (RTD's), two (2) primary pressure transmitters PT402 and PT403. In-core thermocouples.

The in-core thermocouple signals of 0-700°F are converted to 2-10 volts by installing four (4) new E/I convertors.

The above signals are transmitted to a mini computer located in the CCR. The computer compares the signals to a pre-determined set point and sends out an alarm in the CCR to warn the CCR operators that we are approaching flashing, or boiling point in the primary system.

The temperature and pressure signals will be directed to the saturation meter located in the CCR and the readout from this meter will read out on section "D" of the flight panel.

Any malfunction due to the modification will not increase the probability of occurrence or the consequences of any accident or malfunction of equipment important to safety previously evaluated in the safety analysis.

The modification does not create any possibility for an accident or malfunction that has not been previously evaluated and maintains the margins of safety as defined in the Technical Specification.

It is therefore, determined that the modification does not constitute an unreviewed safety question.

21. Cycle 3/4 Refueling

During the Cycle 3/4 refueling, five Region 2 fifty six Region 3 and seven Region 4 fuel assemblies were replaced by sixty eight Region 6 fuel assemblies. The Region 6 assemblies have a nominal enrichment of 3.35 (w/o of U-235) and a nominal overall assembly weight of 649.1 Kg., and the same exterior dimensions as the assemblies now in the core.

A report, entitled "Reload Safety Evaluation, Indian Point Nuclear Plant, Unit 2, Cycle 4" was prepared by Westinghouse and independently reviewed by Con Edison in-house personnel.

The report presents an evaluation for Cycle 3 which demonstrated that the core reload will not adversely affect the safety of the plant. All incidents analyzed and reported in the FSAR which could potentially be affected by the fuel reload were reviewed for the Cycle 4 design. The results of new analyses are included, and the justification for the applicability of previous results for the remaining analyses is presented. It has been concluded that the Cycle 4 design does not cause previously acceptable safety limits for any incident to be exceeded. This conclusion is based on the assumption that: (1) Cycle 3 operation is terminated between 10,550 and 11,350 MWD/MTU, (2) Cycle 4 burnup is limited to the end-of-life full power capability and (3) there is adherence to plant operating limitations given in the technical specifications.

The only concerns relating to reload not specifically addressed by the subject report involve storage and handling of the Region 6 fuel enrichment of 3.5 (w/o U-235). As stated above, the Region 6 fuel has a nominal enrichment of 3.35 (w/o U-235), well below the design value, and does not therefore, raise any additional concerns relating to the technical specification limitation of keff ($\leq$ 0.90). Since the exterior dimensions and configuration of the fuel assemblies are unchanged, they will continue to fit properly into the present fuel racks. Additionally, the nominal fuel assembly total weight for Region 6 is the same as that for Region 5. As a result, the Cycle 3/4 reload does not degrade the fuel handling and storage system's seismic design or normal load bearing capability.

As stated above, the subject report concludes that the Cycle 4 design does not cause the previously acceptable safety limits for any incident to be exceeded. Since safe plant operation is not jeopardized by the creation of any condition below those limits, it can be concluded that the probability and consequences of previously evaluated accidents addressed by that report are not increased. For previously evaluated accidents not specifically addressed by the report (storage and handling) it has been shown that previous design criteria are still valid. The probability and/or consequences of those accidents are, therefore, unaffected.

The methods of plant operation, fuel storage and fuel handling are not affected by this reload. Thus no new type of accident can be created.

Finally, it has been demonstrated that safety limits which dictate the bases for the plant technical specifications will not be exceeded by the Cycle 3/4 reload. Therefore, the margins of safety as defined in those bases are not reduced.

It is concluded that the Cycle 3/4 refueling and Cycle 4 operation of Indian Point Unit #2 does not constitute and unreviewed safety question per 10CFR50.59.

22. <u>Containment Pressure Relief Valves Modification to Pneumatic Control Circuits Valves PCV-1190, 91 & 92</u>

The pressure relief system consists of 3 butterfly valves in series, numbers PCV-1190, 91 & 92. The valves are pneumatically controlled with indication in the CCR for fully open & fully closed positions.

The valves, as they previously existed, had a valve positioner located in the control circuit. Impurities in the control air had been causing the pneumatic positioner to become sluggish & stick, thus, causing the valves operating circuit to fail. The normal operation mode of these valves is either full open or full closed, there is no interim position used.

This modification includes removal of the positioner and a slight redesign of the pneumatic control circuit. The redesign includes the replacement of certain valves & the addition of a filter to aid in removing the impurities which have been determined to be the root cause for past failures.

The three containment pressure relief valves are designed to fail closed. This control change will increase the reliability of the system, thus reducing the probability of valve position failure.

The modification doesn't affect the seismic or containment isolation requirements of the existing valves. In addition there are no security or fire protection changes or requirements necessary due to this modification.

Thus, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report will not be increased.

The valves' positioners, that were removed from the three containment isolation valves, are normally used if a valve position other than full open or full closed is to be used. The valves at Indian Point are only required to be fully open or closed, thus the removal of the valves positioner will have no effect on the day to day normal operation or containment isolation requirements of the valves. Therefore, the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report will not be increased.

The technical specifications do not directly discuss the containment pressure relief valves in any section. But, in section 3.6.B the containment pressure is required to be kept below 2 PSIG. If the containment pressure relief valves are used to meet specification 3.6.B this modification will aid in maintaining the margin of safety as defined in that specifications basis by increasing the reliability of the 3 pressure relief valves. Thus, the margin of safety as defined in the based for any technical specification will not be reduced.

Therefore, this modification is deemed not to involved an unreviewed safety question.

---

23. <u>Inaccessible Hydraulic Snubber Modification</u>

Indian Point Unit No. 2 lines equipped with inaccessible snubbers were analyzed by comparing the same lines in Unit No. 3 for similar arrangement and restraints.

After a detailed comparison (Specification and Analysis Engineering Section), of the Unit 3 lines, it was determined that the IP 2 snubbers could be modified by duplicating the support arrangement on similar lines in Indian Point Unit No. 3. The modifications were broken down into three groups:

Group I - Remove hydraulic snubber completely with no replacement with any other type restraint or additional modification required.

Group II - Modify entire line supports and restraints to duplicate arrangement used on similar line on Indian Point Unit 3.

Group III - Replace identified hydraulic snubber with rigid restraints. For this group, it is intended that a rigid strut be fabricated to replace only the hydraulic snubber section of existing pipe restraint assembly.

The following conditions must be considered in the design of supports and restraints for a Seismic Class I piping system: normal conditions such as deadweight load, thermal expansion & dynamic loads (e.g., water-hammer, safety-valve discharge), & abnormal conditions such as seismic motion & pipe rupture effects. The restraints & supports affected by this modification are designed to control deadweight & seismic loads & thermal expansion stress. The Unit 3 criteria for these conditions are as comprehensive & restrictive as the criteria for Unit 2. The design methods to satisfy those criteria were different however.

In the design of Unit 2 and Unit 3, seismic restraints were located in accordance with span tables. The span tables for Unit 2 were derived by requiring that the natural frequency of pipe spans between restraints be in a low-amplification range of the earthquake response spectra, and that the highest possible response amplitude of the building be assumed with a conservatively selected allowable seismic stress. The span tables for Unit 3 were derived separately for various elevations of the buildings to take advantage of the reduced building response at lower elevations, thus allowing larger separation of restraints compared to Unit 2. Also, for Unit 3, lines 6 inches in diameter and larger and the high-head safety injection lines were verified by dynamic seismic analysis. Separate span tables employing a larger factor of safety were applied to those lines not subject to dynamic seismic analysis. To determine whether a rigid seismic restraint could be used or a hydraulic snubber would be required on the design of Unit 2, the anticipated free thermal displacements of the piping were considered. For Unit 3, on the other hand, thermal stress analyses were performed on all thermally-affected lines permitting more frequent use of rigid restraints. Thus, because additional analytical information was available during the design of Unit 3 piping (building floor response spectra and thermal stress analyses), not only were fewer seismic restraints required on a given line in comparison with the same line on Unit 2, but of those restraints, relatively fewer were required to be of the hydraulic snubber type.

The thermal stresses due to the change from snubber to rigid restraint were taken into consideration when the piping analysis was done from Unit No. 3 and the change to the Unit 3 configuration in Unit 2 will allow all pipe stresses, due to thermal expansion, to remain within acceptable limits.

The changes to the piping supports in Unit 2 to the same configuration as Unit 3 will provide the same protection for postulated events as previously analyzed. With the revised restraints (Unit 3 configuration) analysis (Unit 3) has proven that the piping stresses are within acceptable limits. Thus, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report will not be increased. The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report is not created, inasmuch as the lines in question are not being rerouted or the capacity of the system involved altered. The Unit 3 design, which is applied to Unit 2, has taken the thermal growth into consideration when positioning supports to keep thermal stresses within acceptable design limits. The modification to the inaccessible snubber as described above will not change the ability or capacity of any safety system from completing their design function as described in the Tech. Specs. The supports (as compared to IP3) have been reviewed (Specification and Analysis Section) and will not change the operation (either safety or normal operation) of any system as described in the bases of the present technical specifications.

Thus, the margin of safety as defined in the bases for any technical specifications will not be reduced.

Therefore, the modification is deemed not to involve an unreviewed safety question.